Group Constants Generation of the Pseudo Fission Products for Fast Reactor Burnup Calculations

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Generally burnup calculations of fast reactors are performed with the lumped fission product of each actinide. 172 fission products based on ENDF/B-VI were selected for generation of the pseudo fission product cross sections of each actinide such as U235, U238, Pu239, Pu240, Pu241 and Pu242. The cross sections of 172 nuclides were generated at 400 K and 850 K with the NJOY code. The burnup chains of 172 nuclides were made to use the cumulative or independent fission product yield data for weighting the cross section data of 172 nuclides. The pseudo fission product cross section sets of the 6 actinides were prepared with MATXS-format of 80 energy groups. The MATXS-format cross sections can be easily transformed to the other form such as ISOTXS-format using the TRANSX code. Each lumped cross section includes the elastic, inelastic scattering and capture cross sections.

This paper will describe the generation process of the pseudo fission product cross sections for fast reactor analyses. The burnup chains of 172 nuclides, procedure of the lumping the cross sections and how to use the sets will be included.

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